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Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Docket No. 50-277
SUBJECT: Licensee Event Report, Peach Bottom Atomic Power Station Unit 2

This LER reports a valid Reactor Protection System actuation due to a main turbine trip. Subsequent primary containment isolations occurred due to low reactor water level. The LER is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(iv)(A).

Reference:	Docket No.	50-277
Report Number:	2-01-002	
Revision Number:	00	
Event Date:	07/01/00	
Report Date:	08/30/01	

Facility: Peach Bottom Atomic Power Station Unit 3
1848 Lay Road, Delta, PA 17314-9032

Sincerely,

for GLJ

Gordon L. Johnston, Plant Manager

GLJ/scb

enclosure

cc: PSE&G, Financial Controls and Co-owner Affairs
R. R. Janati, Commonwealth of Pennsylvania
INPO Records Center
H. J. Miller, US NRC, Administrator, Region I
R. I. McLean, State of Maryland
A. C. McMurtray, US NRC, Senior Resident Inspector
A. F. Kirby III, DelMarVa Power

IE22

CCN 01-14083

NRC FORM 366 (1-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2001
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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TITLE (4)

Main Turbine Trip Results in Actuation of the Reactor Protection System.

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
07	01	01	01	0 0 2	00	08	30	01	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)							
			20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
POWER LEVEL (10)	100		20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
			20.2203(a)(1)		50.36(c)(1)(i)(A)	x	50.73(a)(2)(iv)(A)		73.71(a)(4)
			20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
			20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
			20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
			20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
			20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
			20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Steven C. Beck - Regulatory Assurance	(717) 456-3243

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	JJ	JX	G 0 8 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On June 30, 2001 at approximately 1900, the operating crew noticed erratic indications on the main turbine control valve servos, pressure set, and various EHC indicators. I&C was contacted and a troubleshooting plan was developed to determine the cause of the fluctuations. On July 1, 2001, at approximately 0206 hours, an EHC power supply failed and the Unit 2 Main Turbine tripped causing the reactor to automatically scram on a turbine control valve fast closure signal.

As a result of the main turbine trip and reactor scram, reactor pressure increased to approximately 1140 psig and reactor water level decreased to approximately -13 inches. This resulted in four safety relief valves lifting to control reactor pressure and PCIS group II and III isolations occurring, as expected. Additionally, both Reactor Recirculation pumps received a trip signal when reactor pressure exceeded 1106 psig. No Emergency Core Cooling System actuations occurred due to this event. The operating crew stabilized the plant per applicable procedures.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Unit Conditions Prior to the Event

Unit 2 was in Mode 1 and operating at approximately 100% rated thermal power when the event occurred. There were no structures, systems or components out of service that contributed to this event.

Description of the Event

On evening of June 30, 2001, during shift turnover, the operating crew noticed erratic indications on the main turbine control valve servos, pressure set, and various EHC indicators (EIIS:TA). No turbine control valve movement was noted at this time. I&C was subsequently contacted to develop a troubleshooting plan. The plan was developed, approved, and a pre-job brief was conducted.

The initial step in troubleshooting involved obtaining readings from the -22 Volt power supply. The I&C technician noticed that the as-found voltage reading was -22.05 Volts instead of the expected -22.00 Volts. Also, a 59 milliVolt ripple was detected instead of the expected 17 milliVolt ripple. On July 1, 2001, at approximately 0206 hours, shortly after getting the as-found readings, the -22.00 Volt power supply failed (EIIS:JJ). The power supply failure resulted in a Unit 2 Main Turbine trip and caused the reactor to automatically scram on a turbine control valve fast closure signal.

As a result of the main turbine trip and reactor scram, reactor pressure increased to approximately 1140 psig and reactor water level decreased to approximately -13 inches. This resulted in four safety relief valves lifting initially to control reactor pressure and PCIS group II and III isolations occurring, as expected. Reactor pressure was subsequently controlled by the main turbine bypass valves. Additionally, both Reactor Recirculation pumps received a trip signal when reactor pressure exceeded 1106 psig. No Emergency Core Cooling actuations occurred due to this event. All other systems responded as expected for the given plant conditions. The operating crew stabilized the plant per applicable procedures.

Subsequent to the event, the -22 Volt power supply was replaced with a new power supply and all parameters indicated normally. The failed power supply was quarantined for further analysis.

This report is being submitted pursuant to 10CFR50.73(a)(2)(iv)(A) due to a valid actuation of specified systems. The specific systems being reported included a valid actuation of the Reactor Protection System due to a main turbine trip and a valid primary containment isolation of PCIS group II and III valves due to low reactor water level.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Analysis of the Event

A main turbine trip from high power with bypass valves available is analyzed in the UFSAR as an abnormal operational transient (UFSAR Section 14.5.1.2.1). According to the USFAR analysis, the closure of the main turbine stop valves should generate an automatic scram signal and the EHC system should generate a signal to open the main turbine bypass valves. Additionally, the analysis states that reactor vessel safety relief valves open for a short period of time as a result of the initial pressure transient. No fuel damage or unacceptable system stress result from this transient.

After the main turbine trip on July 1, 2001, all systems responded as expected for the transient. The reactor scrammed immediately, four safety relief valves lifted for a short period of time as a result of the initial pressure transient, and main turbine bypass subsequently opened to control reactor pressure. The maximum reactor steam dome pressure was 1140 psig and the minimum reactor water level was -13 inches. Reactor water level was restored to the normal band and controlled by the reactor feedwater system. All mitigating systems operated as expected during this transient.

This event is not a significant operational event. The event was previously analyzed to result in no fuel damage or unacceptable system stress and all mitigated systems responded properly. Additionally, this event resulted in no operations outside the design basis, no major deficiencies having potential generic safety implications, no loss of fuel or barrier integrity, no loss of safety function, no adverse generic implications, no unexplained system interactions, no safety related equipment failures, and no crew performance concerns.

Cause of the Event

The cause of the main turbine trip was a failure of an Electro-Hydraulic Control System power supply, which caused a closure of the main turbine control valves. This resulted in a valid reactor scram signal and a valid primary containment isolation signal.

Subsequent failure analysis of the power supply identified a design deficiency in that a resistor rated at 1 Watt was used in an application with a demand of approximately 0.97 Watts. The resistor overheated, which changed the ohm value, and resulted in both the normal and back-up power supplies indicating erroneously. Both power supplies were replaced and tested satisfactorily.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

Corrective Action Completed

The failed power supply was replaced and tested satisfactorily.

Corrective Actions Planned

None

Previous Similar Occurrences

None